Calculation Cover Sheet

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Axial Source Profile Effect on Waste Package Transporter Shielding

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LIST OF ACRONYMS AND ABBREVIATIONS

ANSI	American National Standards Institute
ANS	American Nuclear Society
B-poly	Borated polyethylene
B-steel	Borated steel
BWR	Boiling Water Reactor
B&W	Babcock & Wilcox
cc	cubic centimeter
CDA	Controlled Design Assumptions Document
CDB	Characteristics Data Base
Ci	Curies
cm	centimeters
CPU	Central Processing Unit
CSCI	Computer Software Configuration Item
DBE	Design Basis Event
DBF	Design Basis Fuel
DTN	Data Tracking Number
EBDRD	Engineered Barrier Design Requirements Document
EDA	Enhanced Design Alternative
fsd	fractional standard deviation
g	grams
hr	hours
kg	kilograms
LA	License Application
LEF	Lower End Fittings
MCNP	Monte Carlo Neutral Particle transport code
MeV	Million Electron Volts
mrem	millirem
MTU	Metric Tons of Uranium
MWd	Megawatt Days
n	neutrons
N/A	Not Applicable
PWR	Pressurized Water Reactor
rem	Roentgen Equivalent in Man
s or sec	seconds
SCS	Software Configuration Secretariat
SQR	Software Qualification Report
SNF	Spent Nuclear Fuel
TBV	To Be Verified
UEF	Upper End Fittings
VA	Viability Assessment
WP	Waste Package
YMP	Yucca Mountain Site Characterization Project
yr	years

1. PURPOSE

The purpose of this scoping calculation is to support preliminary design of the Waste Package (WP) transporter radiation shield configuration. Spent Nuclear Fuel (SNF) is highly radioactive and site personnel must be protected during the period that the WPs are emplaced. Personnel protection is accomplished via a heavily shielded WP transporter that moves the waste from the surface to the emplacement drift. All previous WP transporter shielding calculations have assumed a Design Basis Fuel (DBF) in which the fuel burnup is uniform (e.g. Ref. 7.3, Ref. 7.4, and Ref. 7.12). In reality, SNF burnup varies significantly from one end of the fuel assembly to the other. Since source strengths are dependent upon fuel burnup, a model which varies the fuel burnup along the assembly axis will produce a more accurate depiction of the radiation field surrounding the WP transporter. The objective of this calculation is to determine the need for using the actual axial profile, as opposed to the uniform burnup assumption, in the WP transporter shield design. The scope of the calculation is as follows:

- Determine the impact of axial source term variation on WP transporter contact dose rates.
- Determine appropriate shielding modifications to account for expected dose rate peaking effects.

Consistent with the previous subsurface shielding analyses, this calculation considers the bounding 21 Pressurized Water Reactor (PWR) WP only. The calculation will need to be revised and extended to Boiling Water Reactor (BWR) SNF upon selection of the WP design for the License Application (LA) and availability of the source terms from the WP Operations Group.

2. METHOD

In accordance with Assumption 3.1, radiation source terms for this shielding calculation are based on the data available in the Characteristics Data Base (CDB) (Ref. 7.1, TBV-455). An official source term produced by the WP Group and generated by the SCALE code is currently unavailable. The DBF specified in Section 5.1.1 is used to obtain the source terms for the uniform fuel burnup case, based on a WP thermal design limit of 850 watts per fuel assembly for the large 21 PWR waste package (Ref. 7.2, p. 25). This DBF specification has been used previously in several shielding and Design Basis Event (DBE) analyses (Ref. 7.3, p. 15, Ref. 7.4, p. 12). This calculation uses the DBF and shielding arrangement from the Viability Assessment (VA) reference design (Assumption 3.5). The axial burnup profile is obtained from Ref. 7.10, p. 5.2-2. This is a scoping calculation that will not be used for final design, but which allows for comparison with previous shielding calculations using the assumption of uniform fuel burnup. The results, in terms of the axial source profile effect, are applicable to or useful for other design variations (such as EDA II).

This engineering calculation uses the qualified MCNP4B computer code (Ref. 7.5). The quantity of interest as produced from this code is the dose rate in mrem/hr. This code has been qualified for use in the Yucca Mountain Site Characterization Project (YMP), as per QAP-SI-0/Rev. 1. MCNP4B is a Monte Carlo code capable of treating complex, three-dimensional shielding problems involving scattering, streaming and secondary radiation generation for both neutron and gamma exposures. MCNP4B has been used for this purpose in a previous quality affecting document (Ref. 7.6, p. 10).

Analytical representations are developed to appropriately simulate the WP transporter design configuration. The representations include the radiation sources, geometries, and materials required for use in the calculations with the MCNP4B code.

3. ASSUMPTIONS

The following assumptions are used in this engineering calculation. They are either general assumptions from the Controlled Design Assumptions Document (CDA) (Ref. 7.7) or assumptions specific to this analysis. All of them are covered under TBV-326.

3.1 *Estimation of Spent Nuclear Fuel (SNF) radionuclide inventories for planning purposes and scoping analysis will be based upon the Characteristics Data Base (CDB).* (EBDRD 3.2.3.4.C.4) (Ref. 7.7, p. 4-14)
Rationale: This is a CDA assumption applicable to all radiation shield scoping analyses.
Usage: This assumption is used in Section 5.1.2 to obtain radiation source terms.

3.2 *The waste package contains 21 PWR fuel assemblies as the bounding case.*
Rationale: The 21 PWR waste package design has been used in a previous quality affecting document produced by subsurface radiological design (Ref. 7.4, p. 29).
Usage: This assumption is used in Section 5.1.2 to determine the dimensions of the waste package model and to scale the source terms obtained from the CDB.

3.3 *Material densities and weight fractions used in this calculation are those listed in Table 1.*
Rationale: This material data has been used by subsurface design in previous quality affecting documents (Ref. 7.3, pp. 18-19, Ref. 7.8, p. 12). Note that a fresh fuel composition was used instead of the spent fuel specified in the reference. This is a conservative assumption since fresh fuel yields slightly higher results. The composition was obtained by removing all fission products and actinides from the reference and replacing them with 4.2% enriched fuel.
Usage: This assumption is used in Section 5.1.2 as part of the MCNP4B input.

3.4 *The gamma source spectrum will be limited to photons with energies between 0.45 MeV and 3.0 MeV.*
Rationale: The energy groups in this range make virtually all contributions to the dose rates outside the waste package and transporter (Ref. 7.4, p. 40). Photons with energies less than 0.45 MeV cannot penetrate the thick gamma shield and those above 3.0 MeV are too few in number to make a significant contribution.
Usage: This assumption is used in Section 5.1.2 to obtain the normalized gamma source spectra.

3.5 *The WP design and DBF will be from the VA design.*
Rationale: The axially varying source term effect is applicable to all WP designs and is an intrinsic property of all types of SNF.
Usage: This assumption is used in Section 5.1.1 as part of the MCNP4B input.

3.6 *The neutron spectrum is the same for all fuel burnup and enrichment combinations.*
Rationale: The neutron source is dominated by the spontaneous fission of Cm-244 which has it's own characteristic spectrum.
Usage: This assumption is used in Section 5.1.2 to obtain the neutron source spectrum.

3.7 *The axial burnup profile is the same as that used in Ref. 7.10.*
Rationale: The axial burnup profile is insensitive to small differences in enrichment or burnup.
Usage: This assumption is used in section 5.1.2 to determine the burnup profile.

Table 1. Assumed Material Data For Shielding Analysis[a]

	Fuel Region[b]	WP Inner Barrier	WP Outer Barrier	Air Gap	Fuel Basket	Fuel Plenum	LEF[c]	UEF[d]	Gamma Shield	Neutron Shield	Shield Clad
Material	Smeared	Alloy 22[e]	A516	Air	B-steel	Air	Steel	Steel	A516	B-poly	SS316L
Density (g/cc)	3.0431	8.69	7.832	0.001225	8.0038	0.001225	2.5402	2.4701	7.832	0.92	7.9497
Elemental	**Weight %**										
C	0	0.01	0.22	0	0.043	0	0.072	0.070	0.22	85.94	0.030
P	0	0.02	0.035	0	0.015	0	0.041	0.034	0.035	0	0.045
Si	0	0.08	0.275	0	0.584	0	0.904	0.796	0.275	0	0.75
V	0	0.35	0	0	0	0	0	0	0	0	0
Cr	0.017	22.0	0	0	20.686	0	17.193	18.994	0	0	17.00
Mn	0	0.50	0.90	0	1.336	0	1.809	1.548	0.90	0	2.0
Co	0	2.50	0	0	0	0	0.181	0.150	0	0	0
Fe	0.035	3.0	98.535	0	39.270	0	60.834	56.131	98.535	0	65.545
Ni	0	55.53	0	0	32.505	0	9.044	20.504	0	0	12.00
W	0	3.0	0	0	0	0	0	0	0	0	0
Mo	0	13.0	0	0	2.836	0	0	0.757	0	0	2.50
S	0	0.01	0.035	0	0.030	0	0.027	0.025	0.035	0	0.030
Zr	17.14	0	0	0	0	0	9.356	0	0	0	0
B-10	0	0	0	0	0.077	0	0	0	0	0.278	0
B-11	0	0	0	0	0.351	0	0	0	0	1.223	0
H	0	0	0	0	0	0	0	0	0	12.56	0
O	10.54	0	0	20.0	0	20.0	0.018	0.007	0	0	0
Sn	0.244	0	0	0	0	0	0.133	0	0	0	0
N	0.005	0	0	80.0	0.033	80.0	0.388	0.328	0	0	0.100
Al	0	0	0	0	0.133	0	0	0.124	0	0	0
Cu	0	0	0	0	1.499	0	0	0.037	0	0	0
Ti	0	0	0	0	0.600	0	0	0.223	0	0	0
Nb	0	0	0	0	0	0	0	0.636	0	0	0
Ta	0	0	0	0	0	0	0	0.636	0	0	0
U-235	3.025	0	0	0	0	0	0	0	0	0	0
U-238	68.995	0	0	0	0	0	0	0	0	0	0
Total	100	100	100	100	100	100	100	100	100	100	100

[a] Data in this table is from Ref. 7.3, pp. 18-19, Ref. 7.8, p. 12.
[b] Fresh fuel composition, see Assumption 3.3
[c] Lower End Fittings
[d] Upper End Fittings
[e] HASTELLOY Alloy C-22

4. USE OF COMPUTER SOFTWARE AND MODELS

4.1 SOFTWARE APPROVED FOR QA WORK

MCNP4B (CSCI: 30033 V4B2LV) (Ref. 7.5) is the computer code used to perform the dose rate calculations. MCNP4B was obtained from the Software Configuration Secretariat (SCS) and qualified on CPU # 112111 in accordance with QAP-SI-3/Rev.3 and QAP-SI-0/Rev.1. The use of MCNP4B in this calculation is appropriate per the applications and capabilities of the code and is used within the range of validation in the MCNP4B Software Qualification Report (SQR) (Ref. 7.9).

MCNP4B is a general-purpose neutral particle Monte Carlo computer code used for radiation transport and criticality analysis. MCNP4B was chosen to make this calculation for its ability to simulate three-dimensional environments and model the physics of neutron and photon interactions with matter. Other shielding codes used in previous documents (Ref. 7.4 and Ref. 7.6), such as PATH/V88A, are not capable of performing the coupled neutron-photon reactions necessary to calculate neutron and scattered gamma-ray dose. MCNP4B is an upgraded version of MCNP4A, which has been used extensively for this purpose (Ref. 7.3, p. 13, Ref. 7.4, p. 38).

4.2 OTHER SOFTWARE

The CDB was used to generate radiation source terms. All data provided by the CDB must be considered unqualified per TBV-455.

Microsoft Excel97 was used as a spreadsheet for this engineering calculation. Simple calculations were performed by Excel97 and checked by hand.

4.3 MODELS

No models validated in accordance with AP-3.10Q were used in this calculation.

5. CALCULATION

5.1 INPUT PARAMETERS

5.1.1 Design Basis Fuel (DBF)

Parameters for the VA DBF (Assumption 3.5) PWR spent fuel used as the uniform burnup reference are listed below.

> Characteristics: 4.2% initial enrichment, 48086 MWd/MTU burnup (uniform), and 10 years decay after reactor discharge.
>
> Fuel type: B&W 15x15 Mark B.
>
> Heavy metal loading: 0.464 MTU per fuel assembly.

These DBF parameters are the same as those used in Ref. 7.8, p. 16.

5.1.2 Input Data

Source terms are listed in Tables 2 through 7. Note concerning tabular data; data taken from an external source are represented exactly as they appear in the reference, whereas calculated data (and data from the CDB) are rounded to two decimal places. Also, a shorthand version of scientific notation is used throughout the document (e.g. 1.21E+12 is read as 1.21×10^{12}).

DTN MO98WSTAN00011.000 applies to selected data referenced from previous subsurface radiological design documents. All inputs for this calculation are designated TBV-326. The CDB data are designated TBV-455.

Figure 1 illustrates the variation in fuel burnup along the assembly axis. The graphs in Figure 1 were obtained by interpolating data points from Figure 5.2-1 on page 5.2-2 of Ref. 7.10 and then taking into account slight differences in fuel length between the reference assembly and the DBF used in this calculation. The burnup zones referred to in Table 2 and Table 4 correspond to the histogram "steps" in Figure 1. The reference assembly and the DBF have a slightly different enrichment and burnup, however it is assumed that the axial burnup profile is the same (Assumption 3.7).

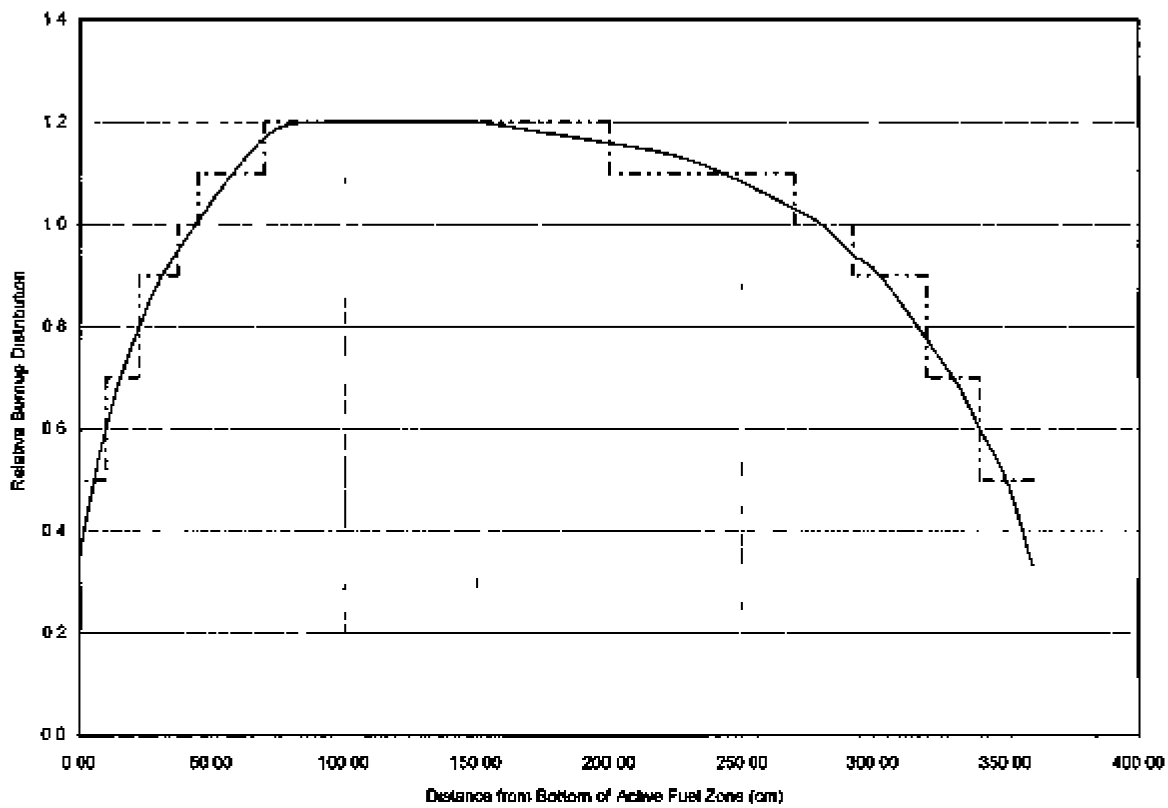

Figure 1. PWR DBF Axial Burnup Profile

Table 2 provides the neutron source strengths for the PWR DBF as a function of distance along the assembly axis. Burnup zone data was taken from Figure 1. Source strength data was obtained from the CDB (Ref. 7.1) in units of n/s-MTU. Note that for an enrichment of 4.20%, the two lowest burnup values (24043 MWd/MTU and 33660 MWd/MTU) were unavailable from the CDB. In these four cases (twice at the fuel assembly bottom and twice at the top), the highest enrichments allowable were used and are noted in parentheses. This is a conservative substitution since the enrichments used are below 4.2% (a lower enrichment with the same burnup yields a larger source term). The relative probabilities are used in the MCNP4B input to determine each source neutron's initial axial zone. Table 2 uses Assumptions 3.1 and 3.2.

Table 2. Neutron Source Strength

Fuel Zone (cm from bottom)	Relative Burnup	Burnup (MWd/MTU)	n/s-MTU	Fraction of Fuel Length	n/s-WP[a]	Relative Probability[b]
340.17 - 360.18	0.5	24043	3.82E+07 (3.42%)	0.056	2.08E+07	2.33E-03
320.16 - 340.17	0.7	33660	1.23E+08 (4.05%)	0.056	6.71E+07	7.51E-03
292.65 - 320.16	0.9	43277	4.18E+08	0.076	3.09E+08	3.46E-02
270.14 - 292.65	1.0	48086	6.74E+08	0.062	4.07E+08	4.56E-02
200.10 - 270.14	1.1	52895	1.02E+09	0.194	1.92E+09	2.15E-01
70.04 - 200.10	1.2	57703	1.47E+09	0.361	5.16E+09	5.78E-01
45.02 - 70.04	1.1	52895	1.02E+09	0.069	6.82E+08	7.64E-02
37.52 - 45.02	1.0	48086	6.74E+08	0.021	1.38E+08	1.55E-02
22.51 - 37.52	0.9	43277	4.18E+08	0.042	1.71E+08	1.91E-02
10.01 - 22.51	0.7	33660	1.23E+08 (4.05%)	0.035	4.19E+07	4.69E-03
0 - 10.01	0.5	24043	3.82E+07 (3.42%)	0.028	1.04E+07	1.16E-03
Total				1	8.93E+09	1
0 – 360.18 (Uniform Case)	1.0	48086	6.74E+08	1	6.57E+09	1

[a] n/s-WP = (n/s-MTU) × (Fraction of Fuel Length) × (0.464 MTU/assembly) × (21 assemblies/WP)
[b] Relative Probability = (n/s-WP) / (Total n/s-WP)

Table 3 gives the neutron spectrum for the PWR DBF. This data was taken directly from page 16 of Ref. 7.3. The spectrum is the same for all fuel burnup and enrichment combinations (Assumption 3.6).

Table 3. Neutron Source Spectrum

Neutron Energy (MeV)	Normalized Spectrum
6.43 - 20.00	0.0183
3.00 - 6.43	0.2099
1.85 - 3.00	0.2341
1.40 - 1.85	0.1310
0.90 - 1.40	0.1767
0.40 - 0.90	0.1924
0.10 - 0.40	0.0376
0.0 - 0.10	0
Total	1

Table 4 provides the fuel gamma source strengths for the PWR DBF as a function of distance along the assembly axis. Burnup zone data was taken from Figure 1. Source strength data was obtained from the CDB (Ref. 7.1) in units of photons/s-MTU. Note that for an enrichment of 4.2%, the two lowest burnup values (24043 MWd/MTU and 33660 MWd/MTU) were unavailable from the CDB. In these four cases (twice at the fuel assembly bottom and twice at the top), the highest enrichments allowable were used and are noted in parentheses. The relative probabilities are used in the MCNP4B input to determine each source photon's initial axial zone. Table 4 uses Assumptions 3.1, 3.2, and 3.4.

Table 4. Fuel Gamma Source Strength[a]

Fuel Zone (cm from bottom)	Relative Burnup	Burnup (MWd/MTU)	photons/s-MTU[b]	Fraction of Fuel Length	photons/s-WP[c]	Relative Probability[d]
340.17 - 360.18	0.5	24043	2.67E+15 (3.42%)	0.056	1.46E+15	2.59E-02
320.16 - 340.17	0.7	33660	3.80E+15 (4.05%)	0.056	2.07E+15	3.68E-02
292.65 - 320.16	0.9	43277	5.02E+15	0.076	3.72E+15	6.61E-02
270.14 - 292.65	1.0	48086	5.66E+15	0.062	3.42E+15	6.07E-02
200.10 - 270.14	1.1	52895	6.28E+15	0.194	1.19E+16	2.11E-01
70.04 - 200.10	1.2	57703	6.91E+15	0.361	2.43E+16	4.32E-01
45.02 - 70.04	1.1	52895	6.28E+15	0.069	4.22E+15	7.50E-02
37.52 - 45.02	1.0	48086	5.66E+15	0.021	1.16E+15	2.06E-02
22.51 - 37.52	0.9	43277	5.02E+15	0.042	2.05E+15	3.66E-02
10.01 - 22.51	0.7	33660	3.80E+15 (4.05%)	0.035	1.30E+15	2.31E-02
0 - 10.01	0.5	24043	2.67E+15 (3.42%)	0.028	7.28E+14	1.30E-02
Total				1	5.63E+16	1
0 – 360.18 (uniform case)	1.0	48086	5.66E+15	1	5.52E+16	1

[a] Gamma rays within the specified energy range only (0.45 MeV to 3.0 MeV) (Assumption 3.4)
[b] Sum of gammas within in the specified energy range, from Tables 5 and 6
[c] Photons/s-WP=(photons/s-MTU) × (Fraction of Fuel Length) × (0.464 MTU/assembly) × (21 assemblies/WP)
[d] Relative Probability = (photons/s-WP) / (Total photons/s-WP)

Tables 5 and 6 provide the spectral data for the 6 burnup values used. This data was taken from the CDB (Ref. 7.1) in units of photons/s-MTU. These tables use Assumptions 3.1, 3.2, and 3.4.

Table 5. Fuel Gamma Source Spectra (24043 MWd/MTU – 43277 MWd/MTU)

Photon Energy (MeV)	Mean Photon Energy (MeV)	24043 MWd/MTU		33660 MWd/MTU		43277 MWd/MTU	
		photons/s-MTU	Normalized Spectrum[a]	photons/s-MTU	Normalized Spectrum	photons/s-MTU	Normalized Spectrum
0.0 - 0.02	0.01	1.48E+15	N/A	2.03E+15	N/A	2.50E+15	N/A
0.02 - 0.03	0.025	3.22E+14	N/A	4.36E+14	N/A	5.33E+14	N/A
0.03 - 0.045	0.0375	3.60E+14	N/A	5.04E+14	N/A	6.44E+14	N/A
0.045 - 0.07	0.0575	2.95E+14	N/A	4.03E+14	N/A	4.97E+14	N/A
0.07 - 0.10	0.085	1.71E+14	N/A	2.36E+14	N/A	2.95E+14	N/A
0.10 - 0.15	0.125	1.39E+14	N/A	2.05E+14	N/A	2.78E+14	N/A
0.15 - 0.30	0.225	1.42E+14	N/A	1.96E+14	N/A	2.42E+14	N/A
0.30 - 0.45	0.375	7.00E+13	N/A	9.44E+13	N/A	1.16E+14	N/A
0.45 - 0.70	0.575	2.36E+15	8.84E-01	3.32E+15	8.74E-01	4.30E+15	8.57E-01
0.70 - 1.0	0.85	1.38E+14	5.17E-02	2.41E+14	6.34E-02	3.87E+14	7.71E-02
1.0 - 1.5	1.25	1.74E+14	6.52E-02	2.41E+14	6.34E-02	3.33E+14	6.63E-02
1.5 - 2.0	1.75	1.44E+12	5.39E-04	2.52E+12	6.63E-04	4.06E+12	8.09E-04
2.0 - 2.5	2.25	5.85E+10	2.19E-05	6.57E+10	1.73E-05	7.31E+10	1.46E-05
2.5 - 3.0	2.75	3.00E+09	1.12E-06	3.88E+09	1.02E-06	5.23E+09	1.04E-06
3.0 - 4.0	3.5	3.74E+08	N/A	4.74E+08	N/A	6.35E+08	N/A
4.0 - 6.0	5	1.57E+06	N/A	5.21E+06	N/A	1.79E+07	N/A
6.0 - 8.0	7	1.81E+05	N/A	6.00E+05	N/A	2.07E+06	N/A
8.0 - 11.0	9.5	2.08E+04	N/A	6.90E+04	N/A	2.37E+05	N/A
Total		5.65E+15	1	7.91E+15	1	1.01E+16	1

[a] Normalized Spectrum = (photon/s-MTU) / (Total photons/s-MTU), for gamma energies between 0.45 MeV and 3.0 MeV only (Assumption 3.4)

Table 6. Fuel Gamma Source Spectra (48086 MWd/MTU – 57703 MWd/MTU)

Photon Energy (MeV)	Mean Photon Energy (MeV)	48086 MWd/MTU		52896 MWd/MTU		57703 MWd/MTU	
		photons/s-MTU	Normalized Spectrum[a]	photons/s-MTU	Normalized Spectrum	photons/s-MTU	Normalized Spectrum
0.0 - 0.02	0.01	2.73E+15	N/A	2.95E+15	N/A	3.16E+15	N/A
0.02 - 0.03	0.025	5.79E+14	N/A	6.22E+14	N/A	6.62E+14	N/A
0.03 - 0.045	0.0375	7.13E+14	N/A	7.79E+14	N/A	8.44E+14	N/A
0.045 - 0.07	0.0575	5.40E+14	N/A	5.79E+14	N/A	6.16E+14	N/A
0.07 - 0.10	0.085	3.23E+14	N/A	3.50E+14	N/A	3.75E+14	N/A
0.10 - 0.15	0.125	3.15E+14	N/A	3.51E+14	N/A	3.85E+14	N/A
0.15 - 0.30	0.225	2.64E+14	N/A	2.85E+14	N/A	3.04E+14	N/A
0.30 - 0.45	0.375	1.26E+14	N/A	1.35E+14	N/A	1.44E+14	N/A
0.45 - 0.70	0.575	4.80E+15	8.48E-01	5.29E+15	8.42E-01	5.78E+15	8.36E-01
0.70 - 1.0	0.85	4.67E+14	8.25E-02	5.49E+14	8.74E-02	6.32E+14	9.15E-02
1.0 - 1.5	1.25	3.85E+14	6.80E-02	4.36E+14	6.94E-02	4.87E+14	7.05E-02
1.5 - 2.0	1.75	4.88E+12	8.62E-04	5.68E+12	9.04E-04	6.47E+12	9.36E-04
2.0 - 2.5	2.25	7.69E+10	1.36E-05	8.04E+10	1.28E-05	8.36E+10	1.21E-05
2.5 - 3.0	2.75	5.94E+09	1.05E-06	6.65E+09	1.06E-06	7.36E+09	1.07E-06
3.0 - 4.0	3.5	7.26E+08	N/A	8.24E+08	N/A	9.30E+08	N/A
4.0 - 6.0	5	2.90E+07	N/A	4.38E+07	N/A	6.33E+07	N/A
6.0 - 8.0	7	3.34E+06	N/A	5.05E+06	N/A	7.30E+06	N/A
8.0 - 11.0	9.5	3.84E+05	N/A	5.80E+05	N/A	8.39E+05	N/A
Total		1.12E+16	1	1.23E+16	1	1.34E+16	1

[a] Normalized Spectrum = (photon/s-MTU) / (Total photons/s-MTU), for gamma energies between 0.45 MeV and 3.0 MeV only (Assumption 3.4)

Table 7 provides the Co-60 activation source data for the PWR fuel assembly end fittings and plenum region for the DBF. This data was taken directly from page 19 of Ref. 7.8 in units of Ci/WP. Co-60 emits two significant photons with energies of 1.17 MeV and 1.33 MeV (Ref. 7.8, p. 19).

Table 7. Co-60 Gamma Source Strength

Region	Ci/WP	photons/s-WP[a]	Relative Probability[b]
Upper End Fittings (UEF)	1877	1.39E+14	3.48E-01
Upper Plenum	970	7.18E+13	1.80E-01
Lower End Fittings (LEF)	2541	1.88E+14	4.71E-01
Total	5388	3.99E+14	1.00

[a] photons/s-WP = (Ci/WP) × (3.7E+10 disintegrations/s-Ci) × (2 photons/disintegration)
[b] Relative Probability = (photons/s-WP) / (Total photons/s-WP)

The flux-to-dose rate conversion factors in Tables 8 and 9 are used to convert the calculated neutron and gamma fluxes to dose rates. These factors are based on the ANSI/ANS-6.1.1-1977 Standard (Ref. 7.11, pp. 4-5).

Table 8. Neutron Flux-to-Dose Rate Conversion Factors

Neutron Energy (MeV)	Conversion Factors (rem/hr)/(n/cm^2-s)
2.50E-08	3.67E-06
1.00E-07	3.67E-06
1.00E-06	4.46E-06
1.00E-05	4.54E-06
1.00E-04	4.18E-06
1.00E-03	3.76E-06
1.00E-02	3.56E-06
1.00E-01	2.17E-05
5.00E-01	9.26E-05
1.0	1.32E-04
2.5	1.25E-04
5.0	1.56E-04
7.0	1.47E-04
10.0	1.47E-04
14.0	2.08E-04
20.0	2.27E-04

Table 9. Gamma-Ray Flux-To-Dose Rate Conversion Factors

Photon Energy (MeV)	Conversion Factors (rem/hr)/(photon/cm^2-s)
0.01	3.96E-06
0.03	5.82E-07
0.05	2.90E-07
0.07	2.58E-07
0.1	2.83E-07
0.15	3.79E-07
0.2	5.01E-07
0.25	6.31E-07
0.3	7.59E-07
0.35	8.78E-07
0.4	9.85E-07
0.45	1.08E-06
0.5	1.17E-06
0.55	1.27E-06
0.6	1.36E-06
0.65	1.44E-06
0.7	1.52E-06
0.8	1.68E-06
1.0	1.98E-06
1.4	2.51E-06
1.8	2.99E-06
2.2	3.42E-06
2.6	3.82E-06
2.8	4.01E-06
3.25	4.41E-06
3.75	4.83E-06
4.25	5.23E-06
4.75	5.60E-06
5.0	5.80E-06
5.25	6.01E-06
5.75	6.37E-06
6.25	6.74E-06
6.75	7.11E-06
7.5	7.66E-06
9.0	8.77E-06
11.0	1.03E-05
13.0	1.18E-05
15.0	1.33E-05

Figure 2 illustrates the variation in source strength along the fuel assembly axis. The neutron and fuel gamma sources are shown relative to the uniform burnup case source strengths (i.e. a relative strength of 1.5 means 1.5 times the uniform burnup case for that particle type). The burnup curve from Figure 1 is reproduced here to provide a reference point for the source strength variation.

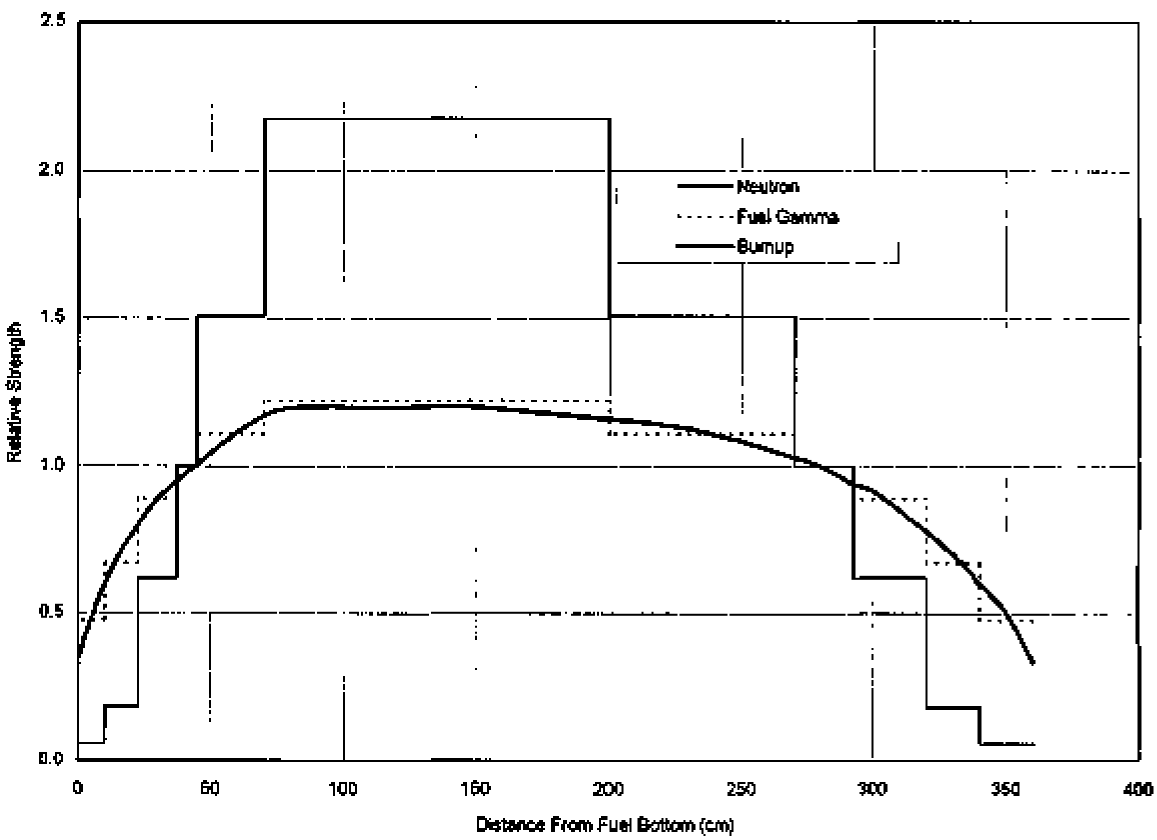

Figure 2. Source Strength Distribution

Table 10 lists the shield thicknesses and materials that comprise the WP transporter. This is designated as the "base configuration" (a modified configuration is presented in Section 6.3). These dimensions are the same as those used in a previous calculation (Ref. 7.3, p. 35, note the conversion from inches to centimeters).

Table 10. WP Transporter Radiation Shield Composition (Base Configuration)

Material	Purpose	Radial Thickness (cm)	Axial Thickness (cm)
SS316L	inner clad	0.50	0.50
A516 Carbon Steel	gamma shield	15.24	17.78
Borated Polyethylene	neutron shield	10.16	7.62
SS316L	outer clad	0.50	0.50
Total		26.40	26.40

Figures 3 and 4 show the geometrical arrangement used to perform the radial and axial dose rate calculations. These drawings are taken from pages 21 and 22 of Ref. 7.6. Figure 4 has been modified to include the upper axial cross-section and has slightly different dimensions.

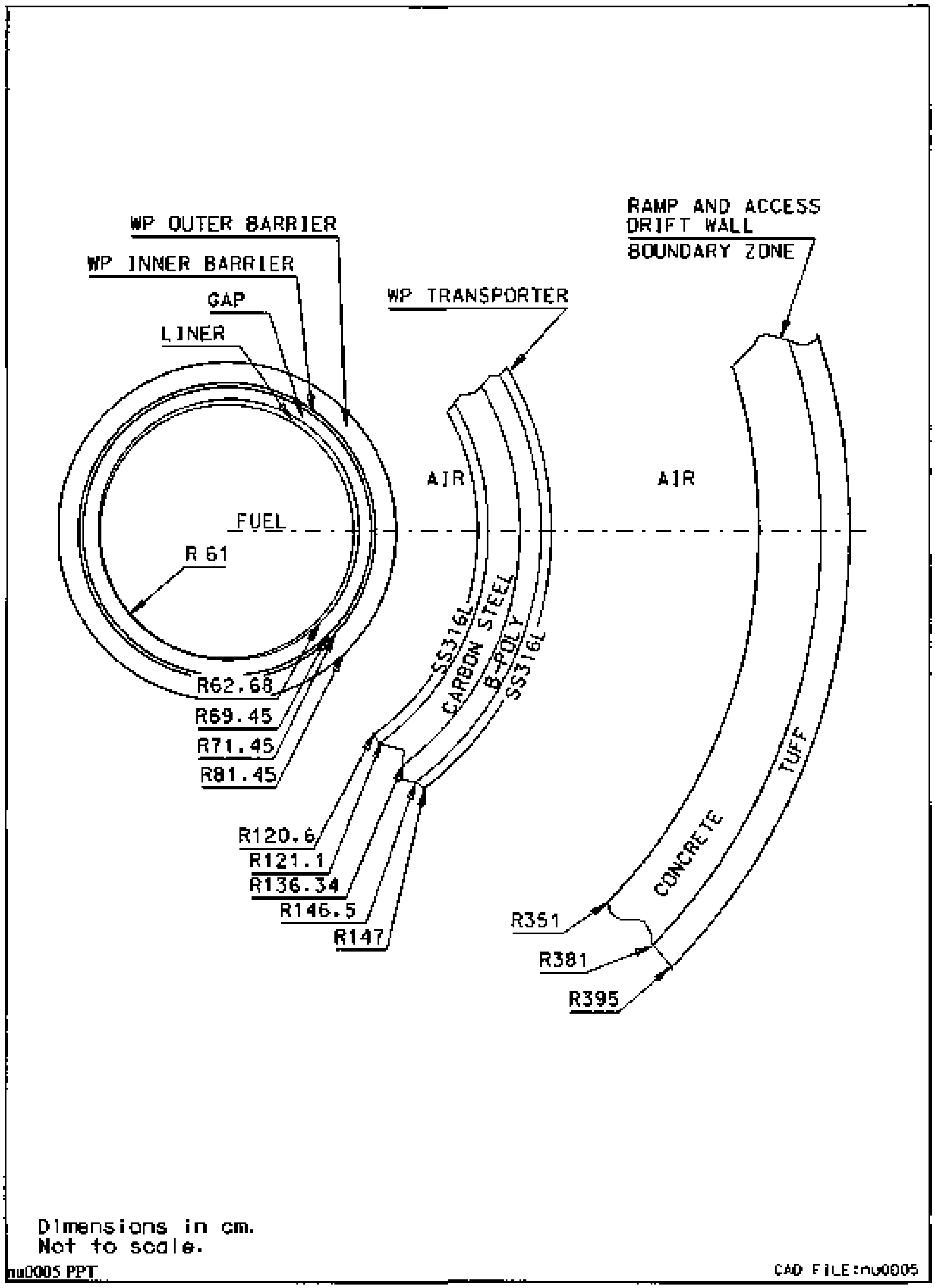

Figure 3. Radial Calculation Geometry

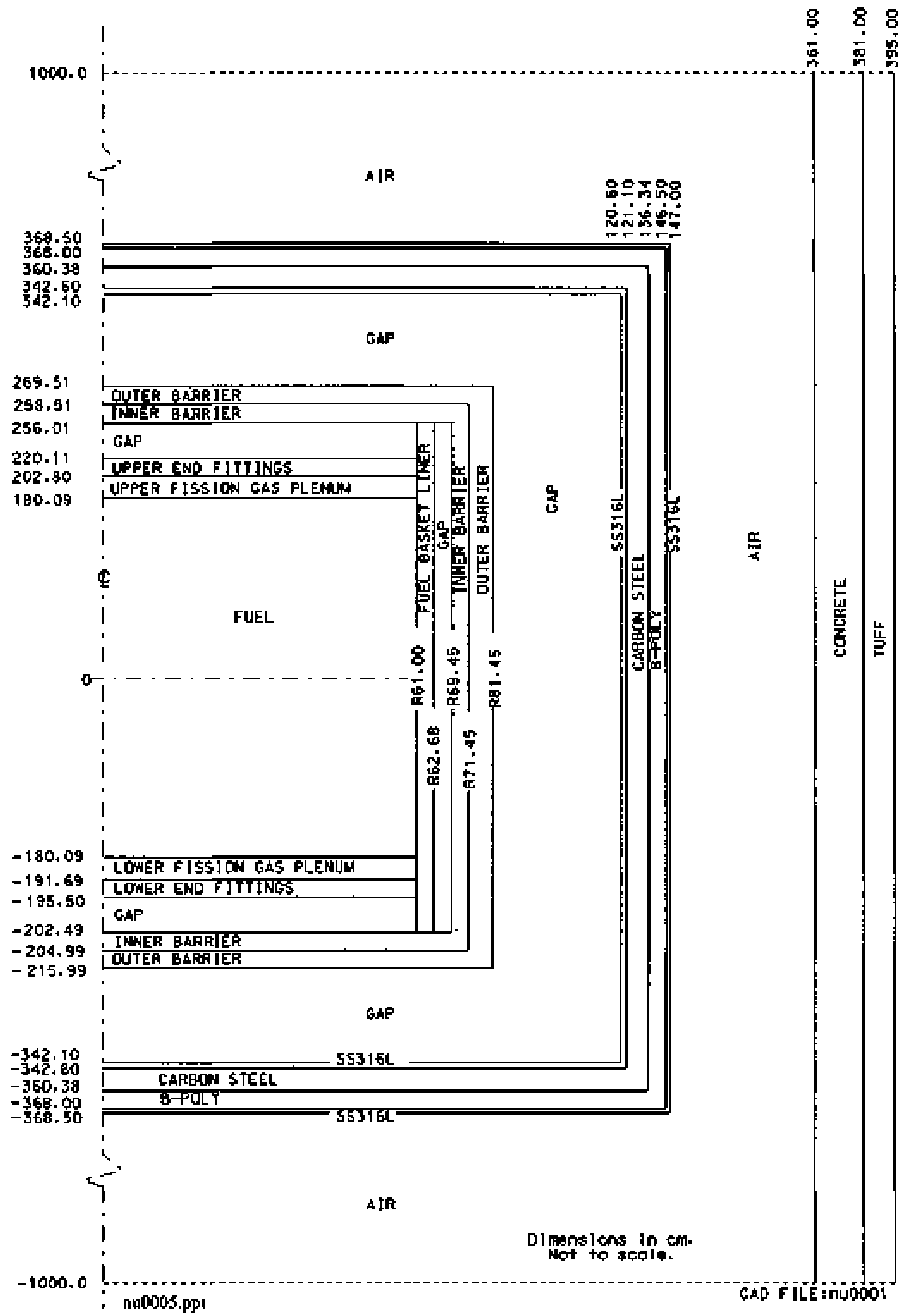

Figure 4. Axial Calculation Geometry[a]

[a] Dimensions in Figure 4 are from the geometric model used in Ref. 7.12

5.2 DOSE RATE CALCULATION

The shielding analysis was performed with the MCNP4B code, including both neutron and gamma radiation fields. For the radial case, three separate calculations were performed: one for the neutron and secondary gamma contributions, one for the primary gamma contributions from the active fuel, and one to include the contribution from the end fitting regions (LEF, UEF, and upper plenum). For the axial case, six calculations were performed, one each of the following for the upper and lower transporter ends: neutron and secondary gammas, primary gammas from fuel, and Co-60 gammas from the end fitting regions. A listing of input and output files is included in Attachment I.

The active fuel region was divided into eleven zones corresponding to different burnup values along the fuel axis (see Table 2). Each zone has a different source strength and (for gamma photons) a different source spectrum (Tables 2 through 6). A segmented surface detector (radial case) was used to record the dose rates corresponding to the various burnup zones. Point detectors were used at the transporter ends (axial case).

To improve statistics, selected biasing schemes were used in the MCNP4B calculations, including source biasing, cell importance specification, and energy splitting biasing. The source biasing consisted of position and energy biasing by placing more particles in source cells closer to the detector and in higher energy bins.

The cell importance was specified by subdividing each major material interval into subcells. Cell importances were specified separately for neutrons and gammas, because of their different attenuation characteristics. A general rule of thumb for the cell importance specification is to increase the cell importance by a factor of 2, when the quantity of interest (such as dose rate) drops by a factor of 2 to 4. This rule provides a good balance of the particle's weight without over-biasing or under-biasing.

Energy splitting was used in the random walk process to play Russian roulette as the particle energy got lower and lower. This biasing improved calculational efficiency by killing the less important particles sooner.

To obtain the absolute results, appropriate normalization factors were included in the MCNP4B input and multiplied by the calculated results which are given on a "per source particle" basis. These factors correspond to the total number of source particles (n/s-WP or photons/s-WP) and are listed in Tables 2, 4 and 7.

6. RESULTS

6.1 DOSE RATE RESULTS

The results of the dose rate calculations, including fractional standard deviations (fsd's) and output filenames, are listed in Tables 11 and 12. Fsd's are due to the statistical nature of Monte Carlo techniques and, when lower than 0.10, are generally considered acceptable levels of error for surface detectors (radial case) (Ref. 7.5, p. 2-95). For point detectors (axial case), fsd's ≤ 0.05 are needed for reliable results (Ref. 7.5, p. 2-95).

Table 11 lists the results of the radial dose rate calculation. The tallying surface for the radial calculation was divided into 17 segments to produce a dose rate profile. All fsd's are below the required 0.10 indicating that these are reliable results. Figure 5 shows the radial dose rates graphically.

Table 12 lists the results of the axial dose rate calculation. Some of the individual fsd's are higher than the required 0.05, however these cases of high fsd's are associated with low dose contributions and are therefore not significant. In every case the total fsd is below 0.05. The fsd for the total result was estimated by the square root of $\Sigma(f_i\sigma_i)^2$, where f_i is the fractional dose contribution from each source component, and σ_i is the associated fsd. Point detectors located on or near surfaces are difficult to deal with as they generally produce tallies with poor statistics (Ref. 7.5, p. 5-50). To minimize this, 5 additional dose points were specified at varying distances from the transporter ends. A smooth curve was fit to this data and the "surface" dose rate listed in Table 12 was actually obtained by extrapolating the curve back to zero distance (i.e. transporter surface). The extrapolation curves are shown in Figures 6 and 7.

Table 11. Radial Dose Rate Results (Base Configuration)

Zone Data		Fuel[b] Gamma (mrem/hr)	Fuel[b] Gamma fsd	^{60}Co[c] Gamma (mrem/hr)	^{60}Co[c] Gamma fsd	Neutron[d] (mrem/hr)	Neutron[d] fsd	Secondary[d] Gamma (mrem/hr)	Secondary[d] Gamma fsd	Total Dose Rate (mrem/hr)	Total fsd
Axial Distance From Transporter Centerline (cm)[a]	Burnup (MWd/MTU)										
269.51 to 368.5	N/A	0.23	0.0482	0.49	0.0490	0.99	0.0498	0.61	0.0382	2.32	0.0261
220.11 to 269.51	N/A	0.81	0.0503	5.09	0.0096	1.65	0.0386	1.16	0.0313	8.71	0.0111
202.80 to 220.11	N/A	2.22	0.0598	13.26	0.0095	2.29	0.0661	1.53	0.0371	19.30	0.0127
180.09 to 202.80	N/A	3.67	0.0321	16.52	0.0081	2.60	0.0345	1.95	0.0335	24.74	0.0085
160.08 to 180.09	24043	6.40	0.0252	11.52	0.0102	3.59	0.0364	2.41	0.0360	23.92	0.0106
140.07 to 160.08	33660	9.32	0.0203	4.19	0.0153	5.28	0.0340	3.04	0.0256	21.83	0.0128
112.56 to 140.07	43277	12.48	0.0167	0.87	0.0254	6.93	0.0384	3.78	0.0211	24.06	0.0145
90.05 to 112.56	48086	14.86	0.0178	0.22	0.0510	8.80	0.0243	4.87	0.0191	28.75	0.0123
20.01 to 90.05	52895	17.59	0.0094	0.13	0.0378	11.73	0.0124	6.21	0.0104	35.66	0.0064
-23.34 to 20.01	57703	19.00	0.0108	0.13	0.0588	14.73	0.0146	7.65	0.0123	41.51	0.0076
-66.70 to -23.34	57703	19.52	0.0115	0.12	0.0512	15.03	0.0144	7.85	0.0122	42.52	0.0077
-110.05 to -66.70	57703	18.93	0.0112	0.16	0.0472	13.55	0.0157	6.83	0.0114	39.47	0.0079
-135.07 to -110.05	52895	16.08	0.0154	0.72	0.0379	10.64	0.0209	5.84	0.0179	33.28	0.0105
-142.57 to -135.07	48086	14.51	0.0228	2.05	0.0275	9.33	0.0417	5.15	0.0272	31.04	0.0172
-157.58 to -142.57	43277	12.09	0.0198	4.16	0.0164	8.33	0.0666	4.56	0.0229	29.14	0.0212
-170.08 to -157.58	33660	9.22	0.0245	7.78	0.0142	6.30	0.0349	3.95	0.0243	27.25	0.0128
-180.09 to -170.08	24043	6.74	0.0304	9.71	0.0137	4.98	0.0355	3.72	0.0526	25.15	0.0143
-191.69 to -180.09	N/A	4.98	0.0348	8.87	0.0124	4.35	0.0428	3.09	0.0327	21.29	0.0139
-195.50 to -191.69	N/A	3.78	0.0585	7.80	0.0169	3.99	0.0547	2.90	0.0517	18.47	0.0200
-215.99 to -195.50	N/A	2.62	0.0476	7.14	0.0118	3.28	0.0325	2.31	0.0284	15.35	0.0128
-368.5 to -215.99	N/A	0.40	0.0478	0.86	0.0251	1.58	0.0220	1.23	0.0210	4.07	0.0128
Centerline (Uniform Burnup)[e]	48086	16.30	0.0163	N/A	N/A	8.30	0.0103	6.40	0.0101	31.00	0.0092

[a] See Figure 4 for axial profile geometry
[b] From output file GRAD.OUT
[c] From output file CORAD.OUT
[d] From output file NRAD.OUT
[e] From Ref. 7.3, pp. 35 & 37

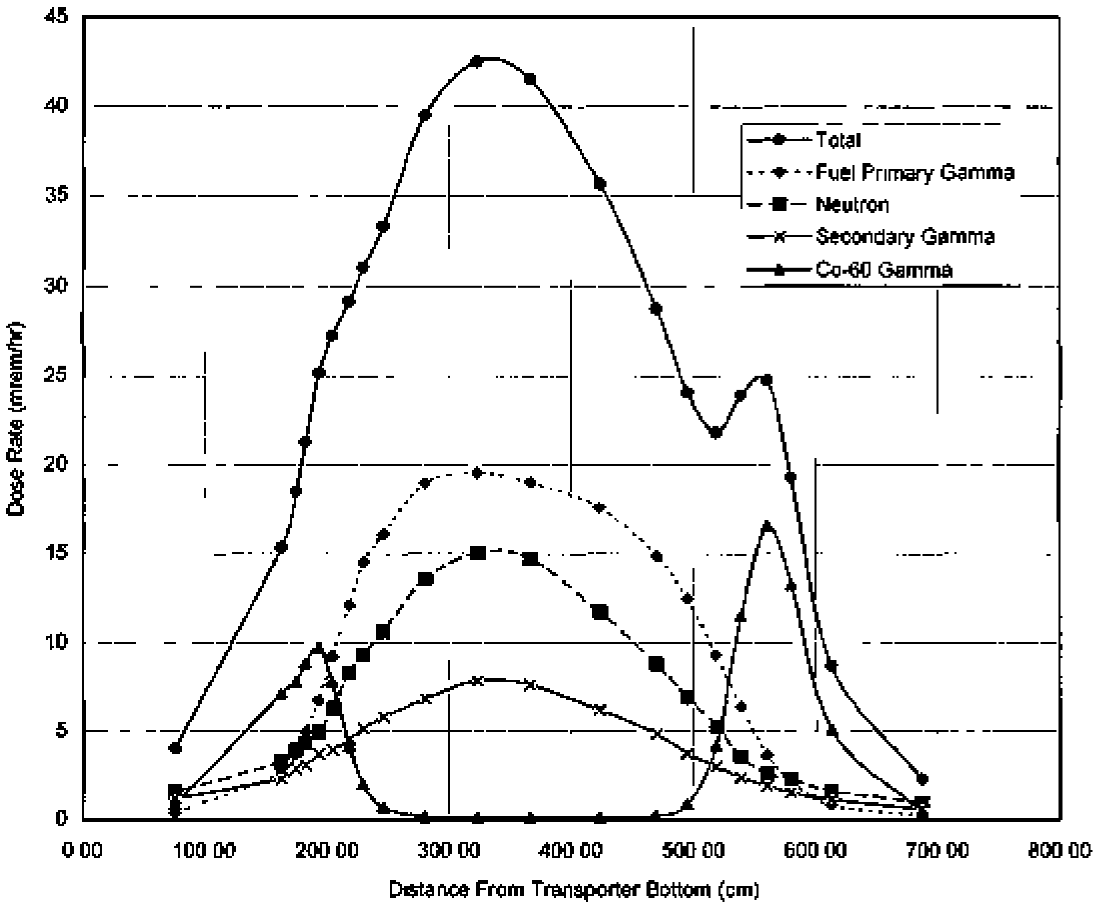

Figure 5. Radial Surface Dose Rates[a]

[a] See Figure 4 for axial profile geometry

Table 12. Axial Dose Rate Results (Base Configuration)

Distance From Surface (cm)	Fuel Gamma (mrem/hr)	Fuel Gamma fsd	^{60}Co Gamma (mrem/hr)	^{60}Co Gamma fsd	Neutron (mrem/hr)	Neutron fsd	Secondary Gamma (mrem/hr)	Secondary Gamma fsd	Total (mrem/hr)	Total fsd
Upper Axial Case[a]										
0 (extrapolated)	0.28	N/A	6.50	N/A	1.32	N/A	1.30	N/A	9.40	N/A
25	0.26	0.1858	4.50	0.0292	1.23	0.0746	1.08	0.0966	7.07	0.0279
50	0.24	0.2060	3.32	0.0191	1.16	0.0420	0.91	0.0641	5.63	0.0197
100	0.19	0.1840	2.03	0.0141	1.04	0.0295	0.63	0.0528	3.89	0.0164
200	0.20	0.2796	1.01	0.0204	0.81	0.0270	0.34	0.0473	2.36	0.0277
400	0.15	0.2435	0.37	0.0165	0.48	0.0262	0.15	0.0334	1.15	0.0343
Lower Axial Case[b]										
0 (extrapolated)	1.81	N/A	11.25	N/A	4.50	N/A	2.85	N/A	20.41	N/A
25	1.30	0.0576	8.29	0.0200	3.76	0.0598	2.26	0.1063	15.61	0.0241
50	1.04	0.0416	6.52	0.0140	3.22	0.0355	1.81	0.0660	12.59	0.0154
100	0.69	0.0338	4.22	0.0104	2.46	0.0238	1.14	0.0512	8.51	0.0113
200	0.42	0.0702	2.07	0.0112	1.57	0.0199	0.56	0.0390	4.62	0.0116
400	0.21	0.0815	0.78	0.0097	0.77	0.0205	0.23	0.0326	1.99	0.0129
0 (Uniform Burnup)[c]	3.34	0.0942	11.31	0.0510	4.18	0.1128	1.72	0.0621	20.56	0.0397

[a] From output files GAX.OUTUP (Fuel Gamma), CAX.OUTUP (Co-60 Gamma), and NAX.OUTUP (n + Secondary Gamma)
[b] From output files GAX.OUTLO (Fuel Gamma), CAX.OUTLO (Co-60 Gamma), and NAX.OUTLO (n + Secondary Gamma)
[c] From Ref. 7.12, p. 45

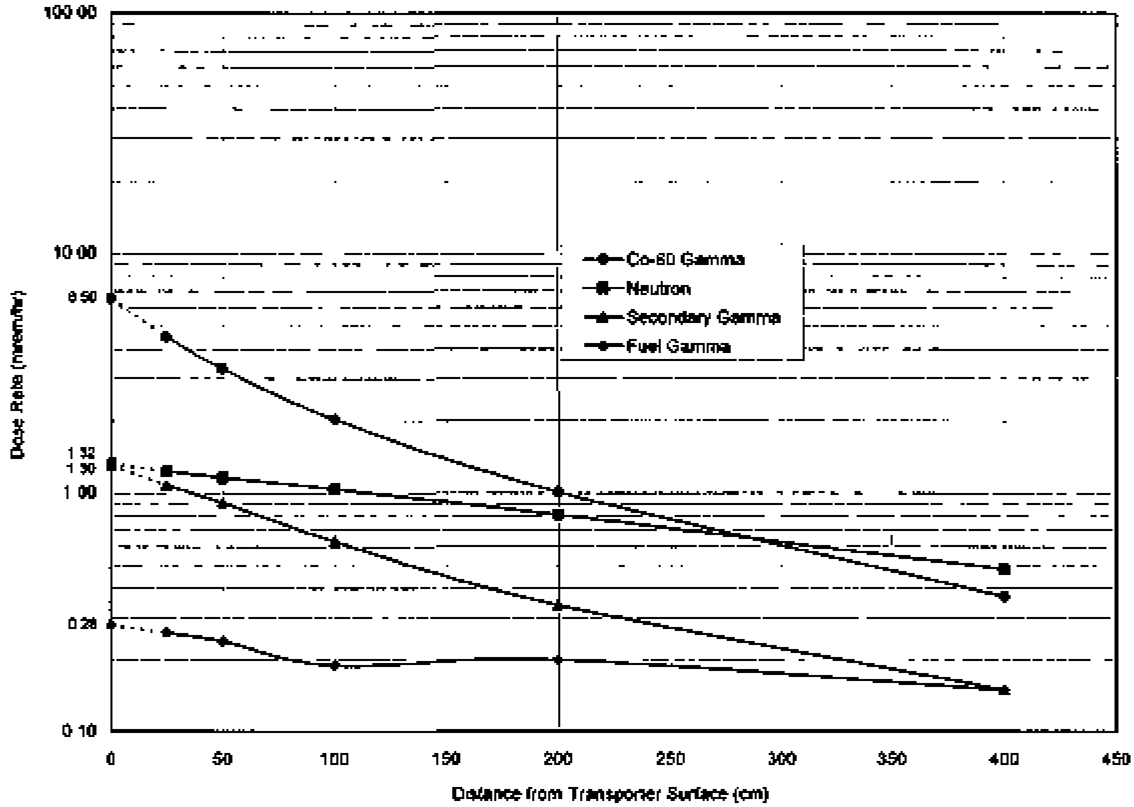

Figure 6. Upper Axial Extrapolated Dose Rates

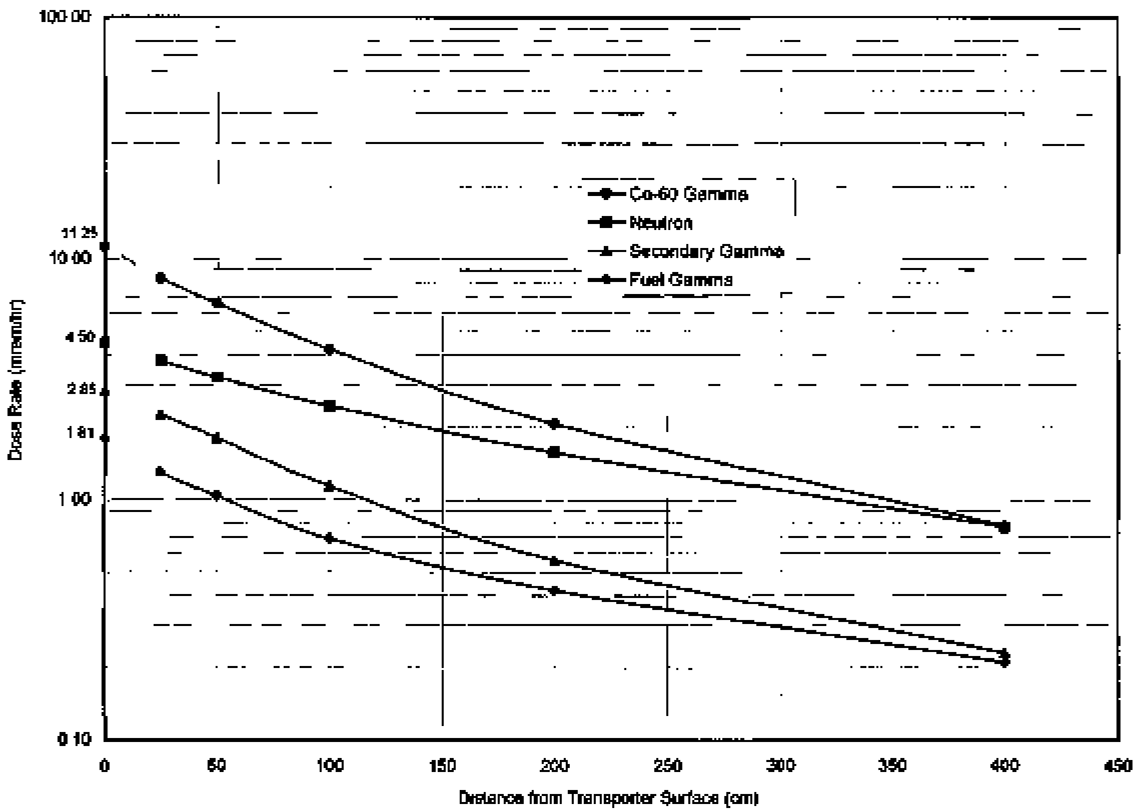

Figure 7 Lower Axial Extrapolated Dose Rates

6.2 COMPARISON WITH UNIFORM BURNUP CASE

Comparisons with previous shielding analyses, in which a uniform fuel burnup distribution was assumed, are contained in Tables 11 and 12. This comparison data was obtained from previous works that used the MCNP4A code and the same techniques as the current calculation.

For the radial case, the most noticeable difference is in the dose rate profile effect of the current data versus the single dose rate value for the uniform case. This is an expected consequence of using source terms based on a varying fuel burnup distribution. The non-uniform case has a peak dose rate higher than the uniform case and then tapers off toward the assembly ends. For the current calculation, the peak dose rate (42.52 mrem/hr) is higher than would be expected by simply applying an axial "peaking factor" to the uniform case dose rate of 31.0 mrem/hr (Ref. 7.3, p. 35). The peaking factor in this case would be 1.2 giving a peak dose rate of 37.2 mrem/hr. This phenomenon can be explained by pointing out that fuel gamma sources are approximately proportional to burnup, but neutron sources are not. The neutron source strength fluctuates almost exponentially with the fuel burnup (see Figure 2). In general, peaking factors are appropriate to use only if the contribution is essentially all from the fuel gammas.

For the axial case, the current calculations are slightly lower than the uniform case. This is to be expected since the neutron and fuel gamma source strengths are weaker toward the assembly ends. Neutron albedo from the high burnup area compensates for the lower source strength and so the compared neutron dose rates are approximately equal. The major dose rate contributor in the axial direction is the Co-60 source in the end fittings and plenum regions. This source is unaffected by fuel burnup variation and the compared dose rates from the LEF are essentially the same. Note that upper axial dose rates from the UEF and upper plenum regions have not previously been calculated, so no comparison can be made.

Assuming a uniform fuel burnup provides results that are slightly conservative in the axial direction, but too low (even with the application of a peaking factor) in the radial direction. This suggests that the axial profile needs to be considered for an accurate determination of the radiation fields surrounding the WP transporter.

6.3 SHIELDING MODIFICATION ESTIMATE

The purpose of the WP transporter is to provide an environment in which radiation dose rates are maintained below an acceptable level during WP emplacement. For repository design purposes, this level has been assumed to be 50 mrem/hr (Ref. 7.4, p. 30) for the outside surface (excluding bottom) of the transporter.

The dose rates in the radial direction (Table 11) are all below the specified limit. As a result, no additional shield material is required for the sides, top, or bottom of the transporter. This appears to be a very efficient design as the peak dose rate is just under the 50 mrem/hr limit.

The dose rates in the axial direction (Table 12) are all below the limit as well, however it may be possible to further optimize the shield design in this region. A new shield arrangement of 15.24 cm (6") carbon steel followed by 7.62 cm (3") borated polyethylene is assumed for the axial direction. This represents a 1" reduction in gamma shielding and no change in neutron shielding.

New dose rates were estimated using the following technique; attenuation factors for each shielding material were obtained by comparing the dose rates from adjacent shield layers using the dose versus depth information on pages 79 and 80 of Ref. 7.4. The factors were then applied to the extrapolated (i.e. surface) axial results to obtain the new dose rates. The attenuation factors are listed in Table 13 and the results are listed in Table 14.

Table 13. Radiation Attenuation Factors[a]

	Shield Material	
Radiation Type	Carbon Steel	B-Poly
Fuel Gamma	2.72	1.18
Co-60 Gamma	2.72	1.18
Neutron	1.33	2.87
Secondary Gamma	1.05	1.28

[a] These are multiplicative factors representing a 1 inch reduction in the given shield material. For example, a 1 inch reduction in the carbon steel gamma shield increases the fuel gamma dose rate by a factor of 2.72.

Table 14. Axial Dose Rate Estimates

	Base Configuration		Modified Configuration	
Shield Thickness (cm)	Lower Axial	Upper Axial	Lower Axial	Upper Axial
Inner Clad	0.50	0.50	0.50	0.50
Gamma Shield	17.78	17.78	15.24	15.24
Neutron Shield	7.62	7.62	7.62	7.62
Outer Clad	0.50	0.50	0.50	0.50
Total	26.40	26.40	23.86	23.86
Dose Rates (mrem/hr)				
Fuel Gamma	1.81	0.28	4.92	0.76
Co-60 Gamma	11.25	6.50	30.60	17.68
Neutron	4.50	1.32	5.99	1.76
Secondary Gamma	2.85	1.30	2.99	1.37
Total	20.41	9.40	44.50	21.57

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7.12 CRWMS M&O 1998. Repository Subsurface Personnel Shielding Design Analysis For Normal Operations. BCAE00000-01717-0200-00003 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980729.0020.

7.13 CRWMS M&O 1999. Computer Files for Axial Source Profile Effect on Waste Package Transporter Shielding, DI: BCAE00000-01717-0210-00005 REV 00. Las Vegas, NV: CRWMS M&O. ACC: MOL.19990420.0438.

8. ATTACHMENTS

There is one attachment to this engineering calculation as described below:

ATTACHMENT I LISTING OF MCNP4B INPUT AND OUTPUT FILES

ATTACHMENT I
LISTING OF MCNP4B INPUT AND OUTPUT FILES AND SAMPLE INPUT FILE

The following is a list of the MCNP4B input and output files. This attachment also contains a sample input file. All files are stored electronically on 3.5" diskettes (Ref. 7.13).

```
 Volume in drive D is DELLWIN95-2
 Volume Serial Number is 1BE3-2E60
 Directory of D:\MCNP4B\axprofile

.             <DIR>        04-20-99 10:09a .
..            <DIR>        04-20-99 10:09a ..
CORAD                11,330  06-08-99  9:14a corad
GRAD                 14,017  06-04-99  3:05p grad
CORAD    OUT        100,938  06-04-99  9:30a corad.out
NAX      LO          13,628  06-16-99  2:03p nax.lo
GAX      LO          27,012  06-16-99  7:52a gax.lo
CAX      LO          13,043  06-16-99  2:27p cax.lo
NAX      UP          13,427  06-22-99  7:51a nax.up
GAX      UP          27,384  06-22-99  7:49a gax.up
CAX      UP          13,669  06-16-99  2:39p cax.up
GRAD     OUT        124,018  06-05-99  7:08a grad.out
NRAD                 12,900  06-04-99  3:00p nrad
NRAD     OUT        138,634  06-04-99 11:07p nrad.out
AXIAL                     0  06-23-99  5:27p axial
NAX~1    OUT        226,620  06-17-99  6:49a nax.outlo
GAX~1    OUT        291,317  06-17-99 10:50p gax.outlo
CAX~1    OUT        253,274  06-19-99 10:53p cax.outlo
NAX~2    OUT        241,071  06-22-99 11:57p nax.outup
GAX~2    OUT        292,774  06-23-99  3:58p gax.outup
CAX~2    OUT        260,846  06-20-99  2:54p cax.outup
        19 file(s)      2,075,902 bytes
         2 dir(s)   1,831,796,736 bytes free
```

Sample Input File NRAD

```
EBS WP transporter radial shielding analysis
c       Neutrons and secondary gammas from fuel
c       non-uniform fuel burnup, 11 different burnup zones along fuel axis
c       segmented surface detector, cell importance bias, radial source bias
c       21 PWR waste package
c       4.2% initial enrichment and 10 years old, average burnup=48086 MWd/MTU
c       waste package transporter in main drift
c       main drift borehole diameter = 7.62 m, concrete lining = 0.3 m
c       ******fuel and radial cell description*********
1       1 -3.0431     -1 32 -101        $active fuel
2       1 -3.0431     -1 101 -102       $active fuel
3       1 -3.0431     -1 102 -103       $active fuel
4       1 -3.0431     -1 103 -104       $active fuel
5       1 -3.0431     -1 104 -105       $active fuel
6       1 -3.0431     -1 105 -108       $active fuel
7       1 -3.0431     -1 108 -109       $active fuel
8       1 -3.0431     -1 109 -110       $active fuel
9       1 -3.0431     -1 110 -111       $active fuel
10      1 -3.0431     -1 111 -112       $active fuel
11      1 -3.0431     -1 112 -20        $active fuel
12      4 -8.0038     1 -2 35 -23       $WP liner
13      5 -0.001225   2 -3 35 -23       $WP gap
14      2 -8.69       3 -4 36 -24       $inner barrier
15      3 -7.832      4 -5 37 -25       $outer barrier (1st 5 cm)
16      3 -7.832      5 -6 37 -25       $outer barrier (2nd 5 cm)
17      5 -0.001225   6 -7 37 -25       $space between WP and transporter
18      6 -7.9497     7 -8 39 -27       $transporter SS inner clad
19      3 -7.832      8 -9 40 -28       $transporter carbon steel gamma shield
20      3 -7.832      9 -10 40 -28      $transporter carbon steel gamma shield
21      3 -7.832      10 -11 40 -28     $transporter carbon steel gamma shield
22      10 -0.92      11 -12 41 -29     $transporter B-poly neutron shield
23      10 -0.92      12 -13 41 -29     $transporter B-poly neutron shield
24      10 -0.92      13 -14 41 -29     $transporter B-poly neutron shield
25      10 -0.92      14 -15 41 -29     $transporter B-poly neutron shield
26      6 -7.9497     15 -16 42 -30     $transporter SS outer clad
27      5 -0.001225   16 -17 42 -30     $space between transporter and drift
28      9 -2.35       17 -18 42 -30     $30 cm concrete lining
29      11 -2.22      18 -19 42 -30     $14 cm into tuff
c       **********axial top cell description***************************
30      5 -0.001225   -1 20 -21         $upper fission gas plenum
31      7 -2.4701     -1 21 -22         $upper end fittings
32      5 -0.001225   -1 22 -23         $gap
33      2 -8.69       -3 23 -24         $inner barrier
34      3 -7.832      -4 24 -25         $outer barrier
35      5 -0.001225   -7 25 -26         $space between WP and transporter
36      6 -7.9497     -7 26 -27         $transporter SS inner clad
37      3 -7.832      -8 27 -28         $transporter carbon steel gamma shield
38      10 -0.92      -11 28 -29        $transporter B-poly neutron shield
39      6 -7.9497     -15 29 -30        $transporter SS outer clad
40      5 -0.001225   -17 30 -31        $air to 10 m from WP center
41      9 -2.35       17 -18 30 -31     $adjacent concrete (30 cm)
42      11 -2.22      18 -19 30 -31     $14 cm into tuff
43      0             19:31:-43         $zero importance
c       **********axial bottom cell description*********
44      5 -0.001225   -1 33 -32         $lower fission gas plenum
45      8 -2.5402     -1 34 -33         $lower end fittings
46      5 -0.001225   -1 35 -34         $gap
47      2 -8.69       -3 36 -35         $inner barrier
48      3 -7.832      -4 37 -36         $outer barrier
49      5 -0.001225   -7 38 -37         $space between WP and transporter
50      6 -7.9497     -7 39 -38         $transporter SS inner clad
51      3 -7.832      -8 40 -39         $transporter carbon steel gamma shield
52      10 -0.92      -11 41 -40        $transporter B-poly neutron shield
53      6 -7.9497     -15 42 -41        $transporter SS outer clad
54      5 -0.001225   -17 43 -42        $air to -10 m from WP center
55      9 -2.35       17 -18 43 -42     $adjacent concrete (30 cm)
56      11 -2.22      18 -19 43 -42     $14 cm into tuff
```

```
c       **********surface description**********
1       cx 61.0         $fuel
2       cx 62.68        $liner
3       cx 69.45        $fuel-wp gap, radial
4       cx 71.45        $wp inner barrier
5       cx 76.45        $wp outer barrier
6       cx 81.45        $wp outer barrier
7       cx 120.6        $transporter ID
8       cx 121.1        $0.5 cm ss316L inner clad
9       cx 126.18       $1st 2" carbon steel
10      cx 131.26       $2nd 2" carbon steel
11      cx 136.34       $3rd 2" carbon steel
12      cx 138.88       $1st 1" b-poly
13      cx 141.42       $2nd 1" b-poly
14      cx 143.96       $3rd 1" b-poly
15      cx 146.50       $4th 1" b-poly
16      cx 147.00       $0.5 cm ss316L outer clad
17      cx 351.0        $concrete lining inner surface
18      cx 381.0        $drift wall surface
19      cx 395.0        $14 cm into tuff
20      px 180.09       $top of fuel
21      px 202.8        $upper plenum
22      px 220.11       $UEF
23      px 256.01       $fuel-wp gap, upper axial
24      px 258.51       $inner barrier, upper axial
25      px 269.51       $outer barrier, upper axial
26      px 342.1        $wp-transporter gap, upper axial
27      px 342.6        $inner clad, upper axial
28      px 360.38       $gamma shield, upper axial
29      px 368.0        $n-shield, upper axial
30      px 368.5        $outer clad, upper axial
31      px 1000.0       $air
32      px -180.09      $bottom of fuel
33      px -191.69      $lower plenum
34      px -195.50      $LEF
35      px -202.49      $fuel-wp gap, lower axial
36      px -204.99      $inner barrier, lower axial
37      px -215.99      $outer barrier, lower axial
38      px -342.1       $wp-transporter gap, lower axial
39      px -342.6       $inner clad, lower axial
40      px -360.38      $gamma shield, lower axial
41      px -368.0       $neutron shield, lower axial
42      px -368.5       $outer clad, lower axial
43      px -1000.0      $air
101     px -170.08      $surfaces 101 - 112 for source specification
102     px -157.58      $and tally segmenting
103     px -142.57
104     px -135.07
105     px -110.05
106     px -66.70
107     px -23.34
108     px 20.01
109     px 90.05
110     px 112.56
111     px 140.07
112     px 160.08

c     active fuel region (fresh fuel)(smear density 3.0431 g/cc)
m1      7014  -0.00005
        8016  -0.10540
       24000  -0.00017
       26000  -0.00035
       40000  -0.17140
       50000  -0.00244
       92235  -0.03025
       92238  -0.68995
c     alloy c-22 (8.69 g/cc)
m2      6000  -0.0001
```

```
      14000  -0.0008
      15031  -0.0002
      16032  -0.0001
      23000  -0.0035
      24000  -0.2200
      25055  -0.0050
      27059  -0.0250
      26000  -0.0300
      28000  -0.5553
      42000  -0.1300
      74000  -0.0300
c     A516 carbon steel (7.832 g/cc)
m3     6000  -0.0022
      14000  -0.00275
      15031  -0.00035
      16032  -0.00035
      25055  -0.0090
      26000  -0.98535
c     liner (SS316B6A + A825, 8.0038 g/cc)
m4     5010  -0.00077
       5011  -0.00351
       6000  -0.00043
       7014  -0.00033
      13027  -0.00133
      14000  -0.00584
      15031  -0.00015
      16032  -0.00030
      22000  -0.00600
      24000  -0.20686
      25055  -0.01336
      26000  -0.39270
      28000  -0.32505
      29000  -0.01499
      42000  -0.02836
c     air (0.001225 g/cc)
m5     7014  -0.8
       8016  -0.2
c     ss316L for transporter clad (7.9497 g/cc)
m6     6000  -0.00030
       7014  -0.00100
      14000  -0.00750
      15031  -0.00045
      16032  -0.00030
      24000  -0.17000
      25055  -0.02000
      26000  -0.65545
      28000  -0.12000
      42000  -0.02500
c     upper end fittings (2.4701 g/cc)
m7     6000  -0.00070
       7014  -0.00328
       8016  -0.00007
      13027  -0.00124
      14000  -0.00796
      15031  -0.00034
      16032  -0.00025
      22000  -0.00223
      24000  -0.18994
      25055  -0.01548
      26000  -0.55131
      27059  -0.00150
      28000  -0.20504
      29000  -0.00037
      41093  -0.00636
      42000  -0.00757
      73181  -0.00636
c     lower end fittings (2.5402 g/cc)
```

```
m8       6000  -0.00072
         7014  -0.00388
         8016  -0.00018
        14000  -0.00904
        15031  -0.00041
        16032  -0.00027
        24000  -0.17193
        25055  -0.01809
        26000  -0.60834
        27059  -0.00181
        28000  -0.09044
        40000  -0.09356
        60000  -0.00133
c       ordinary concrete (2.35 g/cc--ANSI/ANS 6.4-1985)
m9       8016  -0.4983
         1001  -0.0055
        11023  -0.0170
        12000  -0.0026
        13027  -0.0455
        14000  -0.3157
        16032  -0.0013
        19000  -0.0191
        20000  -0.0826
        26000  -0.0123
c       neutron shield - 1.5% B-Polypropylene (0.92 g/cc per GA-4 cask)
m10     1001  -0.1256
        5010  -0.00278
        5011  -0.01223
        6000  -0.8594
c       dry tuff material (2.22 g/cc)
m11     8016  -0.49863
        11023  -0.02909
        12000  -0.00077
        13027  -0.06513
        14000  -0.36898
        15031  -0.00004
        19000  -0.02641
        20000  -0.00322
        22000  -0.00056
        25055  -0.00035
        26000  -0.00682
c       **********Source specification, Biasing, and Tallies*************
mode  n p
imp:n 1 12r 1.5 2 3r 3 4 8 32 128 256 512 2r 128 32 0.5 12r 0 0.5 12r
imp:p 0.5 26r 2 8 0.5 12r 0 0.5 12r
esplt:n 0.5 1.0-3 0.5 1.0-4 0.25 1.0-5 0.25 1.0-6
esplt:p 0.5 0.5 0.5 0.25 0.5 0.1
phys:n  25.0 1.0-7
phys:p  20.0
sdef     cel=d1 pos=-180.09 0 0 ext=fcel d2 erg=d14 rad=d15 axs=1 0 0 par=1
c        axial source positioning
si1  L   1  2  3  4  5  6  7
         8  9  10 11
sp1      1.16e-3     4.69e-3      1.91e-2
         1.55e-2     7.64e-2      5.78e-1
         2.15e-1     4.56e-2      3.46e-2
         7.51e-3     2.33e-3
ds2  s   3           4            5
         6           7            8
         9           10           11
         12          13
si3      0 10.01
sp3      0 1
si4      10.01 22.51
sp4      0 1
si5      22.51 37.52
sp5      0 1
```

```
si6     37.52 45.02
sp6     0 1
si7     45.02 70.04
sp7     0 1
si8     70.04 200.10
sp8     0 1
si9     200.10 270.14
sp9     0 1
si10    270.14 292.65
sp10    0 1
si11    292.65 320.16
sp11    0 1
si12    320.16 340.17
sp12    0 1
si13    340.17 360.18
sp13    0 1
c       unbiased neutron spectrum
si14    0.10  0.40  0.90  1.40  1.85  3.00  6.43  20.00
sp14    0.000 0.038 0.192 0.177 0.131 0.234 0.210 0.018
c       biased radial source distribution
si15     0.0  10.0   20.0   30.0   40.0   50.0   61.0
sp15     0.0  0.0269 0.0806 0.1344 0.1881 0.2419 0.3281
sb15     0.0  0.01   0.03   0.06   0.10   0.20   0.60
fc12  Segmented surface detector, 19 zones, from bottom to top
      neutron dose rates
      ANSI 6.1.1-1977 (mrem/hr)
f12:n 16
fs12  -43 -42 -37 -34 -33 -32 -101 -102 -103 -104 -105 -106
      -107 -108 -109 -110 -111 -112 -20 -21 -22 -25 -30 t
sd12  1.00000E-05 1.00000E-05 1.40863E+05 1.89251E+04 3.51902E+03 1.07141E+04
      9.24552E+03 1.15454E+04 1.38637E+04 6.92721E+03 2.31092E+04 4.00393E+04
      4.00485E+04 4.00393E+04 6.46909E+04 2.07909E+04 2.54090E+04 1.84818E+04
      1.84818E+04 2.09756E+04 1.59880E+04 4.56272E+04 9.14300E+04 1.00000E-05
      6.80714E+05
e12   6.25-7 0.1 1.0 25.0
de12  2.5-8  1.0-7  1.0-6  1.0-5  1.0-4
      1.0-3  1.0-2  1.0-1  5.0-1  1.0
      2.5    5.0    7.0    10.0   14.0   20.0
df12  3.67-3 3.67-3 4.46-3 4.54-3 4.18-3
      3.76-3 3.56-3 2.17-2 9.26-2 1.32-1
      1.25-1 1.56-1 1.47-1 1.47-1 2.08-1 2.27-1
fm12  8.93E+09
fc32  Segmented surface detector, 19 zones, from bottom to top
      gamma dose rates
      ANSI 6.1.1-1977 (mrem/hr)
f32:p 16
fs32  -43 -42 -37 -34 -33 -32 -101 -102 -103 -104 -105 -106
      -107 -108 -109 -110 -111 -112 -20 -21 -22 -25 -30 t
sd32  1.00000E-05 1.00000E-05 1.40863E+05 1.89251E+04 3.51902E+03 1.07141E+04
      9.24552E+03 1.15454E+04 1.38637E+04 6.92721E+03 2.31092E+04 4.00393E+04
      4.00485E+04 4.00393E+04 6.46909E+04 2.07909E+04 2.54090E+04 1.84818E+04
      1.84818E+04 2.09756E+04 1.59880E+04 4.56272E+04 9.14300E+04 1.00000E-05
      6.80714E+05
e32   0.045 0.1 0.45 0.70 1.0 1.5 2.0 3.0
fm32  8.93E+09
de32  0.010   0.030   0.050   0.070   0.100   0.150
      0.200   0.250   0.300   0.350   0.400   0.450
      0.500   0.550   0.600   0.650   0.700   0.800
      1.000   1.400   1.800   2.200   2.600   2.800
      3.25    3.75    4.25    4.75    5.0     5.25
      5.75    6.25    6.75    7.5     9.0     11.0
      13.0    15.0
df32  3.96-3  5.82-4  2.90-4  2.58-4  2.83-4  3.79-4
      5.01-4  6.31-4  7.59-4  8.78-4  9.85-4  1.08-3
      1.17-3  1.27-3  1.36-3  1.44-3  1.52-3  1.68-3
      1.98-3  2.51-3  2.99-3  3.42-3  3.82-3  4.01-3
      4.41-3  4.83-3  5.23-3  5.60-3  5.80-3  6.01-3
      6.37-3  6.74-3  7.11-3  7.66-3  8.77-3  1.03-2
      1.18-2  1.33-2
```

```
fq0   e g
print 310 50
ctme  480
```